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Indiana Michigan Power
Cook Nuclear Plant
One Cook Place
Bridgman, MI 49106
IndianaMichiganPower.com

January 15, 2016

AEP-NRC-2015-92
10 CFR 50.73

Docket No.: 50-316

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Donald C. Cook Nuclear Plant Unit 2
LICENSEE EVENT REPORT 316/2015-001-01
Manual Reactor Trip Due To A Secondary Plant Transient

In accordance with 10 CFR 50.73, Licensee Event Report (LER) System, Indiana Michigan Power Company, the licensee for Donald C. Cook Nuclear Plant Unit 2, is submitting as an enclosure to this letter the following report:

LER 316/2015-001-01: "Manual Reactor Trip Due To A Secondary Plant Transient"

There are no commitments contained in this submittal.

Should you have any questions, please contact Mr. Michael K. Scarpello, Regulatory Affairs Manager, at (269) 466-2649.

Sincerely,

Quinton A. Es

Shane Q. Lies
Site Vice President

JEN/ams

Enclosure

c: R. J. Ancona - MPSC
A. W. Dietrich – NRC Washington, DC
MDEQ – RMD/RPS
NRC Resident Inspector
C. D. Pederson – NRC Region III
A. J. Williamson - AEP Ft. Wayne

IE22
NRR

Enclosure to AEP-NRC-2015-92
LER 316/2015-001-01
Manual Reactor Trip Due To A Secondary Plant Transient

NRC FORM 366 (11-2015) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 10/31/2018

LICENSEE EVENT REPORT (LER)

(See Page 2 for required number of digits/characters for each block)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Donald C. Cook Nuclear Plant Unit 2	05000316	1 OF 5

4. TITLE

Manual Reactor Trip Due To A Secondary Plant Transient

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
04	23	2015	2015 -	001	- 01	01	15	2016	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)			
2	☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
	☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
	☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(B)
	☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
10. POWER LEVEL	☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
002	☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
	☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(C)	☐ 73.77(a)(1)
	☐ 20.2203(a)(2)(v)	☒ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(D)	☐ 73.77(a)(2)(i)
	☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(vii)	☐ 73.77(a)(2)(ii)
		☐ 50.73(a)(2)(i)(C)	☐ OTHER	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

LICENSEE CONTACT	TELEPHONE NUMBER (Include Area Code)
Michael K. Scarpello, Regulatory Affairs Manager	(269) 466-2649

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
D	SB	FCV	C635	Y					

14. SUPPLEMENTAL REPORT EXPECTED	15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES (If yes, complete 15. EXPECTED SUBMISSION DATE) ☒ NO				

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

On April 23, 2015, at 0210, Donald C. Cook Nuclear Plant Unit 2 Reactor was manually tripped from approximately 2 percent of rated thermal power during plant restart following a refueling outage. Unit 2 Reactor was manually tripped due to the inability to maintain Average Reactor Coolant System Temperature above the Technical Specification (TS) required minimum Temperature for Criticality when two newly installed Steam Dump Valves failed open while being manually valved into service. The valves were subjected to, but not designed for, two phase flow.

The Root Cause has been determined to be that the modification process failed to identify and document all system operational vulnerabilities. The corrective action to preclude repetition is an enhancement of the Engineering Modifications procedure to require development and inclusion of a narrative to describe system operation, including key interfacing system operation.

The manual Reactor Protection System (RPS) actuation was reported via Event Notification 51004 in accordance with 10 CFR 50.72(b)(2)(iv)(B) and 10 CFR 50.72(b)(3)(iv)(A), and 10 CFR 50.72(b)(2)(i). The valid RPS actuation and the completion of the plant shutdown required by TS are reportable as a Licensee Event Report in accordance with 10 CFR 50.73(a)(2)(iv)(A) and 10 CFR 50.73(a)(2)(i)(A) respectively.

NRC FORM 366A (11-2015) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104 EXPIRES: 10/31/2018
LICENSEE EVENT REPORT (LER) CONTINUATION SHEET	Estimated burden per response to comply with this mandatory collection request: 80 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the FOIA, Privacy and Information Collections Branch (T-5 F53), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to Infocollects.Resource@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

1. FACILITY NAME	2. DOCKET	3. LER NUMBER YEAR	SEQUENTIAL NUMBER	REV NO.
Donald C. Cook Nuclear Plant Unit 2	05000- 316	2015	- 001	01

NARRATIVE

Energy Industry Identification System codes are identified in the text as [XX].

INTRODUCTION

On April 23, 2015, at 0210, Donald C. Cook Nuclear Plant Unit 2 Reactor [RCT] was manually tripped from Mode 2 at approximately 2 percent (%) of rated thermal power during plant restart following a refueling outage. Unit 2 was manually tripped due to the inability to maintain Average Reactor Coolant System (RCS) [AB] Temperature (Tavg) above the Technical Specification (TS) required minimum Temperature for Criticality when two newly installed Condenser [SG] Steam Dump Valves [SB] [FCV] failed open while being manually valved into service. The valves were subjected to, but not designed for, two phase flow.

EVENT DESCRIPTION

Following heat-up of the Main Steam [SB] system while coming out of a refueling outage, Operators were in the process of restoring flowpaths that were isolated to accommodate the heatup. Isolation valves for two of the three Group I Steam Dump Valves (one of which was one of the newly installed valves) were open with the Steam Dump System modulating as expected. With the isolation valve for the third of the three Group I Steam Dump Valves approximately 17% open, while waiting for steam flow to equalize, operators noted that the position indicating arm on the Steam Dump Valve being placed into service (another one of the newly installed valves) appeared to be fluctuating. The Steam Dump Valve then failed open such that the indicating arm traveled beyond the upper travel (open) limit switch. The field operators communicated to the control room operators that they could hear high steam flow through one of the first two valves that had been placed into service indicating that it was also failed open. The field operators were directed to close the manual isolation valves [ISV] for the failed open Steam Dump Valves. Operators in the control room attempted to close the Steam Dump Valves using the controller, but were not successful. The valve of previous design (not replaced during the outage) operated correctly, the newly installed valves did not respond.

The increase in steam demand resulted in lowering RCS Tavg. The Unit Supervisor immediately directed a manual reactor trip when Tavg lowered from approximately 547 degrees Fahrenheit (°F) to below the minimum temperature for criticality, 541°F, and then below the 539°F operating limit established in the Reactor Start-Up procedure. RCS cooldown was terminated at 522.7°F Tavg by closing the manual isolation valves for the Steam Dump Valves following the trip.

Plant parameters were normal at the onset of the event – RCS pressure was approximately 2235 pounds per square inch gauge, Tavg was approximately 547°F, and there were no components or systems inoperable that contributed to the event.

All control rods fully inserted following the manual reactor trip. Prior to and following the event, Steam Generator (S/G) levels were maintained using the two Motor Driven Auxiliary Feeedwater (MDAFW) pumps feeding all four S/Gs.

Both MDAFW pumps had been placed in service prior to the event and continued to operate under manual operator control in accordance with plant procedures following the Reactor trip. S/G levels did not lower to the automatic actuation setpoint for either the Turbine Driven Auxiliary Feedwater (TDAFW) or the MDAFPs during the transient.

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Secondary Heat Sink was maintained during the transient by feeding all four S/Gs using the Unit 2 MDAFW pumps with steam relief capability available with S/G Power Operated Relief Valves [SB] [PCV] (PORV). Additionally, the Group I Steam Dump Valve which was not replaced during the outage was in service before the event and was controlling RCS temperature normally. Although available, neither the TDAFW pump nor the Main Feedwater [SJ] Pumps were required in order to maintain secondary heat sink. Plant electrical safety busses [BU] were powered from the preferred offsite electrical power source [EB] before and after the manual reactor trip.

The manual Reactor Protection System (RPS) actuation was reported via Event Notification 51004 in accordance with 10 CFR 50.72(b)(2)(iv)(B) and 10 CFR 50.72(b)(3)(iv)(A), and 10 CFR 50.72(b)(2)(i). The valid RPS actuation and the completion of the plant shutdown required by TS are reportable as a LER in accordance with 10 CFR 50.73(a)(2)(iv)(A) and 10 CFR 50.73(a)(2)(i)(A) respectively.

EVENT ANALYSIS

A support/refute analysis refuted all causes except the presence of water in the bottom of the valve flashing to steam causing sudden backpressure and outward thrust forces that exceeded the actuator's capabilities. This resulted in the valves failing to the full open position.

COMPONENT

10-inch Copes Vulcan (SPX) Severe Duty class 600 Generation III tandem trim design air operated control valves.

ASSESSMENT OF SAFETY CONSEQUENCES

NUCLEAR SAFETY
Actual Impact

Failure of the two Steam Dump Valves resulted in minimal nuclear safety impact. All control rods fully inserted as a result of the manual reactor trip, and were unaffected by the RCS cooldown. The reduction in RCS Tavg did not result in a loss of required shutdown margin or in an uncontrolled return to criticality during the post-trip response. RCS heat sink was maintained using the S/Gs with make-up from the MDAFW pumps and steam relief to the atmosphere via S/G PORVs.

Potential impact

Resulting from a RCS cooldown, the following are potential nuclear safety impacts:
- Inability to maintain the Reactor subcritical following shutdown
- Loss of effectiveness of ex-core nuclear instrument [DET] trip setpoint effectiveness
- Exceeding Pressurizer [PZR] TS thermal stress limits
- Inability to maintain the condenser as a secondary heat sink
- Loss of RCS pressure control due to inability to maintain Pressurizer level

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INDUSTRIAL SAFETY

Actual Impact

There was no actual industrial safety hazard resulting from the internal mechanical failure of the Steam Dump Valves. No failure of the Main Steam or Main Condenser pressure boundaries occurred during this event.

Potential Impact

The magnitude of the forces involved in the mechanical transient was sufficient to challenge the valve actuator, but was not sufficient to challenge the steam piping pressure boundary. As long as the valve and actuator remained bolted together, an external steam leak or release would not be expected.

RADIOLOGICAL SAFETY

Actual Impact

There was no actual radiological safety hazard resulting from the internal mechanical failure of the Steam Dump Valves.

Potential Impact

The potential failure of the ability to isolate a failed steam dump flow path could result in the inability to control Main Steam pressure. During conditions such as a Steam S/G [SG] Tube Rupture where the Main Condenser remains available, the loss of steam pressure control would necessitate isolation of the Main Steam lines. This would require RCS cooldown using S/G Power Operated Relief Valves. Potential dose resulting from this cooldown would remain bounded by the dose analysis.

PROBABILISTIC RISK ASSESSMENT (PRA)

A PRA risk assessment was performed on the event. The analysis concluded that the Incremental Conditional Core Damage Probability and Incremental Conditional Large Early Release Probability are under the Regulatory Guide 1.174 criteria for significant events. Therefore, this event was considered to be of very low risk.

ROOT CAUSE

The Root Cause has been determined to be that the modification process failed to identify and document all system operational vulnerabilities.

CORRECTIVE ACTIONS

Immediate Corrective Action Taken

The two newly installed Steam Dump Valves that failed open (of the three installed during the outage) were removed and replaced with the originally installed valves.

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A Temporary Modification has been implemented to allow operation throughout the remainder of the operating cycle with the remaining Copes Vulcan Steam Dump Valve isolated.

Corrective Action to Preclude Repetition (CATPR)

The CATPR is a revision of the Engineering Modifications procedure to require development and inclusion of a narrative to describe system operation, including key interfacing system operation.

Additional Corrective Actions

Planned

Conduct a study to determine the best methodology to preclude water being introduced into the steam dump system through design and procedural changes.

Taken

Revise PMP-5040-MOD-007 (Engineering Modifications) to require a Risk Assessment be performed in accordance with 12-EHP-2291-RIS-001 (Engineering Risk Analysis) and provided to management prior to management approval of modifications being planned and conducted under compressed time schedules.

PREVIOUS SIMILAR EVENTS

Licensee Event Reports (LER) for both units for the past three years were reviewed for similar events related to 10 CFR 50.73(a)(2)(i)(A) reporting criteria for plant shutdown required by TS, and 10 CFR 50.73(a)(2)(iv)(A) reporting criteria for system actuation. The following event was identified:

LER-316-2013-001-00 "Unit 2 Manual Reactor Trip due to Lowering Steam Generator Level"
On July 28, 2013, the Donald C. Cook Nuclear Plant Unit 2 reactor was operating at 100% power . At 1018, reactor operators manually tripped the reactor when reaching a low S/G level threshold during a secondary plant transient event.